NON-PUBLIC?: N
ACCESSION #: 9004300271
LICENSEE EVENT REPORT (LER)

FACILITY NAME: PLANT HATCH, UNIT 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000366

TITLE: INADEQUATE PROCEDURE CAUSES REACTOR SCRAM AND GROUP II ISOLATION
EVENT DATE: 03/28/90 LER #: 90-003-00 REPORT DATE: 04/27/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 3/28/90 at approximately 0815 CST, Unit 2 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% of rated thermal power). At that time, a plant Instrument and Controls (I&C) technician was valving-in pressure transmitter 2E32-NO50 following a routine calibration per procedure 57CP-CAL-103-2S, "ITT Barton Model 764 Differential Pressure Transmitter." Valving-in the pressure transmitter, which shares a common sensing line with Reactor Protection System (RPS) and Primary Containment Isolation System (PCIS) water level transmitters 2B21-N080C and D, caused a pressure perturbation in the sensing line which caused water level transmitters 2B21-N080C and D to spike downscale. This resulted in a false low water level signal being sent to the A2 and B2 trip channels in the RPS and the PCIS. Full scram and partial Group II PCIS isolation signals were generated per design; the

unit scrammed and the outboard Group II Primary Containment Isolation Valves (PCIVs) closed.

The cause of this event is an inadequate procedure. Procedure 57CP-CAL-103-2S did not provide adequate instructions to prevent the pressure perturbation that resulted when pressure transmitter 2E32-N050 was valved-in. Furthermore, the procedure did not provide adequate return to service instructions (e.g., open links, install jumpers, lift leads) necessary to prevent a scram and PCIS actuation in the event of a perturbation in the common sensing line.

Corrective actions for this event include changing procedure 57CP-CAL-103-2S to correct these and similar errors, and reviewing and changing, as necessary, comparable Unit 1 and Unit 2 procedures.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code xx).

SUMMARY OF EVENT

On 3/28/90 at approximately 0815 CST, Unit 2 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% of rated thermal power). At that time, a plant Instrument and Controls (I&C) technician was valving-in pressure transmitter 2E32-N050 following a routine calibration per procedure 57CP-CAL-103-2S, "ITT Barton Model 764 Differential Pressure Transmitter." Valving-in the pressure transmitter, which shares a common sensing line with Reactor Protection System (RPS, EIIS Code JC) and Primary Containment Isolation System (PCIS, EIIS Code JM) water level transmitters 2B21-N080C and D, caused a pressure perturbation in the sensing line which caused water level transmitters 2B21-N080C and D to spike downscale. This resulted in a false low water level signal being sent to the A2 and B2 trip channels in the RPS and PCIS. Full scram and partial (outboard) Group II PCIS isolation signals were generated per design; the unit scrammed and the outboard Group II Primary Containment Isolation Valves (PCIVs, EIIS Code BD) closed.

The cause of this event is an inadequate procedure. Procedure 57CP-CAL-103-2S did not provide adequate instructions to prevent the pressure perturbation that resulted when pressure transmitter 2E32-N050 was valved-in. Furthermore, the procedure did not provide adequate return to service instructions (e.g., open links, install jumpers, lift leads) necessary to prevent a scram and PCIS actuation in the event of a perturbation in the common sensing line.

Corrective actions for this event include changing procedure 57CP-CAL-103-2S to correct these and similar errors, and reviewing and changing, as necessary, comparable Unit 1 and Unit 2 procedures.

DESCRIPTION OF EVENT

On 3/28/90 at approximately 0645 CST, Unit 2 was in the Run mode at an approximate power level of 2436 CMWT (approximately 100% of rated thermal power). At that time, a plant I&C technician began the routine calibration of pressure transmitters 2E32-N050, 2E32-N058, and 2E32-N060 per procedure 57CP-CAL-103-2S. These three pressure transmitters are part of the Main Steamline Isolation Valve (MSIV) Leakage Control System (EIIS Code BF). They provide pressure permissive signals to the logic of the MSIV Leakage Control System to allow the system to be actuated by the operator following a design basis loss-of-coolant accident.

At approximately 0815 CST, the technician had completed the calibration of pressure transmitter 2E32-N050 and was returning it to service per the requirements of the calibration procedure. When he valved-in the instrument, a pressure perturbation was created in the instrument's sensing line. This is a common sensing line shared by several instruments, including water level transmitters 2B21-N080C and D. These instruments provide water level inputs to trip channels A2 and B2, respectively, in the RPS and the PCIS.

The perturbation in the common sensing line caused RPS and PCIS water level transmitters 2B21-N080C and D to spike downscale. When these instruments spiked downscale, they transmitted a false low water level signal to trip units 2B21-N680C and D. The false lo
water level signal
was below the trip units' setpoint of approximately 12 inches above instrument zero; therefore, the trip units actuated resulting in seal-in trips in the A2 and B2 trip channels of the RPS and the PCIS. The one-out-of-two-taken-twice logic of the RPS was satisfied and the unit scrammed per design. Likewise, the, minimum actuation logic for the outboard Group II PCIS was satisfied and the outboard Group II PCIVs closed per design.

Immediately following the scram, actual reactor water level decreased from void collapse to approximately 23 inches below instrument zero (to approximately 11.8 feet above top of active fuel). This is an expected and normal occurrence and resulted in another full scram signal and a full, i.e., inboard and outboard, Group II PCIS isolation signal on low water level approximately six seconds after receipt of the false low water level signal. All four RPS and PCIS trip channels, Al, A2, Bl, and B2, tripped as a result of the actual low water level and the inboard Group II PCIVs closed.

Reactor water level wag restored to its normal level of approximately 36 inches above instrument zero at approximately 0852 CST using the Reactor Feedwater Pumps (RFPs, EIIS Code SJ). Neither the High Pressure Coolant Injection (EIIS Code BG) system nor the Reactor Core Isolation Cooling (EIIS Code BN) system automatically started or injected as water level never reached their automatic initiation setpoint of 35 inches below instrument zero (minimum water level during this event was approximately 23 inches below instrument zero). These two systems were not used in the manual mode during this event because the RFPs were available and sufficient for water level recovery and control.

Reactor vessel pressure was controlled with the Turbine Bypass Valves (EIIS Code SO). Consequently, the Safety Relief Valves (EIIS Code JE) were neither used nor needed during this event. Peak reactor vessel pressure was approximately 985 psig which is normal operating pressure.

At approximately 0855 CST, the unit was in a stable condition and entering the Hot Shutdown mode with reactor water level being maintained at the normal level of approximately 36 inches above instrument zero.

CAUSE OF THE EVENT

The cause of this event is a less than adequate procedure. Procedure 57CP-CAL-103-2S did not identify pressure transmitter 2E32-N050 as an instrument sharing a common sensing line with RPS and PCIS instruments ("pressure boundary instruments"). Consequently, the I&C technician was not required by the procedure to pressurize the instrument to process (i.e., reactor) pressure before valving it into service. This is required by the procedure for those instruments identified as pressure boundary instruments in order to minimize the disturbance to other instruments on the sensing line. Because the instrument was not pressurized to process pressure (it was at approximately 0 psig when it was valved-in), a large pressure perturbation was created in the common sensing line when the instrument isolation valves were opened. This caused spiking in the water level transmitters resulting in a scram and a partial Group II PCIS isolation.

In addition, the procedure's return to service instructions for this instrument were less than adequate. Return to service instructions for other pressure boundary instruments contain steps (e.g., open links, lift leads, install jumpers) to prevent Engineered Safety Feature actuations in the event of a disturbance in a common sensing line. The return to service instructions for this instrument did not contain those steps necessary to prevent trips in both the A2 and B2 RPS and PCIS trip channels. As a result, the instrument spiking caused by the pressure perturbation actuated the logic necessary to cause a scram and a partial Group II PCIS isolation.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required per 10 CFR 50.73(a)(2)(iv) because an unplanned actuation of Engineered Safety Features (ESFs) occurred. Specifically, the RPS actuated on a false low water level signal. The outboard Group II PCIVs of the PCIS, an ESF, closed on the same false low water level signal. Approximately six seconds later, the RPS and PCIS logic systems actuated again and the inboard Group II PCIVs closed, this time on an actual low water level signal as voids collapsed from the scram.

The RPS provides timely protection against the onset and consequences of conditions, such as low water level, that could threaten the integrity of the fuel barriers and the nuclear system process barrier. A reactor scram initiated by a low water level condition protects the fuel by reducing the fission heat generation within the core. The PCIS provides timely protection against the onset and consequences of events involving the potential release of radioactive materials from the fuel and nuclear system process barriers by isolating appropriate lines which penetrate the primary containment. Isolation of Group II PCIVs, initiated by a low water level condition, prevents the escape of radioactive materials from the primary containment through process lines which may have been breached. Additionally, isolation of these process lines conserves reactor coolant inventory if a breach of one of these lines caused the low water level condition.

In the event described in this LER, the RPS and the PCIS actuated and the outboard Group II PCIVs closed on a false low water level signal. Although no actual low water level condition existed, these systems functioned as if one did. All automatic functions of these two systems performed per design. Furthermore, when voids collapsed following. the scram causing an actual low water level condition (a normal occurrence), the RPS and PCIS logic actuated again and the inboard Group II PCIVs isolated per design.

Water level was restored to normal using the RFPs. Water level was never less than approximately 11.8 feet above the top of the active fuel and, in fact, never got low enough to actuate any emergency core cooling systems. Based on this information, it is concluded that this event had no adverse impact on nuclear safety. The above analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Procedure 57CP-CAL-103-2S will be changed to require instruments to be pressurized to process pressure before they are valved into service. This will minimize the possibility of adverse system responses which might lead to an ESF actuation. The return to service instructions for pressure transmitter 2E32-N050 also will be changed to include those steps necessary to prevent a scram and PCIS actuation in the event of a disturbance in its sensing line as the instrument is being valved-in. Additionally, the procedure's return to service instructions for other instruments will be reviewed to ensure they are adequate to prevent ESF actuations. Changes will be made as needed. The review will be completed and the changes made before the procedure is used again.

Likewise, Unit 2 procedure 57CP-CAL-104-2S, "ITT Barton Model 763 Pressure Transmitter," and Unit 1 procedures 57CP-CAL-103-1S, "ITT Barton Model 764 Differential Pressure Transmitter," and 57CP-CAL-104-1S, "ITT Barton Model 763 ,Pressure Transmitter," will be changed to require instruments to be pressurized to process pressure before they are valved into service. These three procedures return to service instructions also will be reviewed to ensure they are adequate to prevent ESF actuations and changes will be made as needed. The reviews will be completed and the changes made before the procedures are used again.

ADDITIONAL INFORMATION

1. Previous Similar Events:

There was one similar event in which the reactor scrammed from a false low water level signal. That event was reported in LER 50-321/1988-002, dated 5/6/88. In that event, I&C personnel were backfilling the reference leg of pressure transmitter 1C32-N004B as part of corrective maintenance activities. That reference leg is shared by RPS and PCIS water level transmitters 1B21-N080C and D. The action of backfilling the common reference leg caused level transmitters 1B21-N080C and D to transmit a false low water level signal to A2 and B2 trip channels, respectively, of the RPS and PCIS logic. As a result, the reactor scrammed and the outboard Group II PCIVs isolated. The corrective actions for that event would not have prevented this event because the causes of the two events are different. The previous event was caused by personnel error whereas this event was caused by an inadequate procedure.

2. Failed Component Identification:

There were no failed components involved in this event. Pressure transmitter 2E32-N050 was being calibrated as part of routine instrument calibration activities and not as part of corrective maintenance activities.

3. Other Affected Equipment:

No systems other than RPS and PCIS were affected by this event.

ATTACHMENT 1 TO 9004300271 PAGE 1 OF 1

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April 23, 1990

U.S. Nuclear Regulatory Commission
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PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
INADEQUATE PROCEDURE CAUSES REACTOR
SCRAM AND GROUP II ISOLATION

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv),
Georgia Power Company is submitting the enclosed Licensee Event Report
(LER) concerning the unanticipated a
tuation of some Engineered Safety
Features (ESFs). This event occurred at Plant Hatch - Unit 2.

Sincerely,

W. G. Hairston, III

JJP/ct

Enclosure: LER 50-366/1990-003

c: (See next page.)

ATTACHMENT 1 TO 9004300271 PAGE 2 OF 2

Georgia Power

U.S. Nuclear Regulatory Commission
April 23, 1990
Page Two

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GO-NORMS

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